

# SUB-CRITICAL TRANSMUTATION REACTORS WITH TOKAMAK FUSION NEUTRON SOURCES

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## ABSTRACT

The principal results of a series of design scoping studies of sub-critical fast transmutation reactors (based on the nuclear and processing technology being developed in the USDoE Generation IV, Advanced Fuel Cycle and Next Generation Nuclear Plant programs) coupled with a tokamak fusion neutron source (based on the ITER design basis physics and technology) are presented.

## I. INTRODUCTION

For many years there has been a substantial R&D activity devoted to closing the nuclear fuel cycle. During the 1990s this activity emphasized the technical evaluation of reducing the requirements for long-term geological high-level waste repositories (HLWRs) for the storage of spent nuclear fuel (SNF) by transmutation (fission) of the plutonium and higher transuranics in the spent fuel discharged from fission power reactors<sup>1-8</sup>. Recycling of this SNF in thermal spectrum fission power reactors, the most obvious option, was found to not significantly reduce the HLWR requirements<sup>1,2</sup>, because the destruction of transuranics (by neutron fission) would be offset by the production of more transuranics by neutron capture transmutation of the isotope <sup>238</sup>U that constitutes about 95% of (slightly enriched) thermal reactor fuel. Repeated recycling of the SNF in special purpose fast spectrum reactors was found to be more effective, but with the net destruction rate of transuranics still limited by the requirement for the presence of <sup>238</sup>U to provide a negative reactivity coefficient for safety and by a safety-related limit on the transuranics loading. There is a potential to relax these two safety-related limits if the reactor is operated sub-critical, with a neutron source making up the neutron deficit to sustain the neutron chain reaction. A general consensus emerged from these studies that significantly higher transuranics net destruction rates could be achieved in sub-critical reactors<sup>1,2</sup>.

The accelerator community was quick to recognize the opportunity to use a D+ accelerator with a spallation target as a neutron source for a sub-critical transmutation reactor. Almost all of the studies in the 1990s of sub-critical transmutation

reactors were based on an accelerator-spallation neutron source<sup>1-8</sup>. In the USA, these studies and the supporting R&D development were organized by DoE under the Accelerator Transmutation of Waste (AWTR) Program<sup>6</sup>, which has now evolved into the Advanced Fuel Cycle Initiative<sup>9</sup> (AFCI).

The USDoE Generation IV (GEN-IV) nuclear reactor development activity<sup>10</sup> envisions that the pacing item for the development of a transmutation reactor--the development of the spent fuel processing technology--should be sufficiently advanced by about 2020 that the detailed design of a critical fast transmutation reactor and the associated processing facility could be started, which would enable the entire system to be brought online in about 2030. The roadmap<sup>6</sup> for developing sub-critical transmutation reactors driven by accelerator-spallation neutron sources also envisions such a reactor coming online in about 2030.

A sub-critical transmutation reactor (using the same nuclear and separations technology) driven by a tokamak fusion neutron source could be brought online somewhat later. The pacing items in bringing online a tokamak neutron source to drive a sub-critical transmutation reactor would be the operation of ITER (or a similar facility) as a prototype and the operation of a set of fusion technology test facilities needed to develop component reliability. ITER is scheduled to operate from 2015 to 2035. Component test facilities could be upgraded or constructed to operate before and in parallel with ITER, so it would be plausible to begin detailed design of a tokamak neutron source in about 2025. Construction of a sub-critical reactor using the same fast reactor technology developed for critical reactors and a tokamak fusion neutron source could then begin as early as about 2030, leading to initial operation in about 2040.

The fusion community has been rather slower in examining the opportunity of using a fusion neutron source for a sub-critical transmutation reactor, with only a few studies<sup>11-16</sup> through the end of the 1990s. Since that time we have undertaken at Georgia Tech a series of studies<sup>17-24</sup> of coupling a tokamak fusion neutron source based largely on ITER design basis physics and technology<sup>25</sup> with a sub-critical transmutation reactor based on the nuclear and processing technology being developed

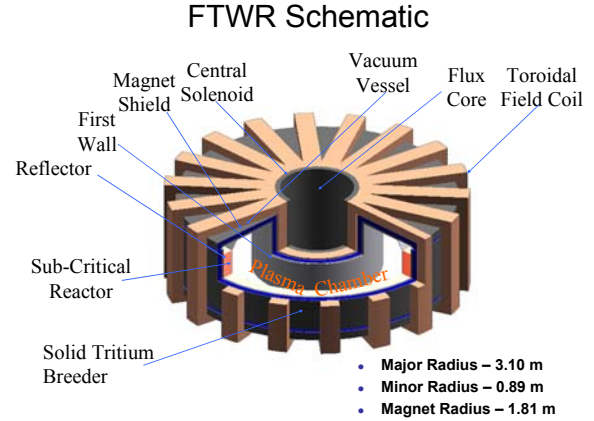
in the USDoE GEN-IV, AFCI and NGNP programs<sup>9,10,26</sup>.

## II. THE FTWR AND GCFTR STUDIES

We have examined sub-critical transmutation reactors based on two of the nuclear technologies being developed in the GEN-IV studies. The Fusion Transmutation of Waste Reactor (FTWR) series of studies was based on a fast-spectrum reactor using a metal fuel consisting of TRU (transuranics) alloyed with zirconium in a zirconium matrix and cooled by a liquid metal (Li17Pb83 eutectic), which also served as the tritium breeder. The ongoing Gas Cooled Fast Transmutation Reactor (GCFTR) series of studies is based on a fast-spectrum reactor using TRU-oxide fuel in coated TRISO particle form in a SiC matrix cooled by He. Both the FTWR and GCFTR cores are annular and located outboard of the toroidal plasma chamber. The core plus plasma chamber were surrounded first by a reflector and then by a shield to protect the magnets from radiation damage and heating, as indicated in Fig. 1 for the initial FTWR design.

A design objective was to use near-term nuclear technology being developed in the DoE Nuclear Program (GEN-IV, AFCI, NGNP) and near-term fusion technology being developed in the ITER Project. The ANL metal fuel, liquid metal cooled reactor designs<sup>8</sup> were adapted to accommodate a different coolant and TRU-Zr fuel for the FTWR designs. The fast, gas-cooled reactor designs being developed under the GEN-IV Program guided the

choice of the GCFTR core design, and the coated fuel particle technology being developed in the NGNP program<sup>26</sup> was adapted to accommodate TRU-oxide fuel for the GCFTR.



**Fig. 1 Tokamak Fusion Transmutation of Waste Reactor**

The fusion technology was based on the ITER design<sup>25</sup>. The superconducting magnet design was based directly on the ITER superconducting magnet system. The first-wall and divertor designs were based on adaptations of the ITER designs to accommodate different coolants. The reference materials compositions for the FTWR and GCFTR designs are given in Table I.

**TABLE I Reference Materials Composition of FTWR and GCFTR**

Component	FTWR	GCFTR
Reactor		
Fuel	TRU-Zr metal in Zr matrix	TRU TRISO/SiC matrix (option BISO/Zirc-4 matrix)
Clad/structure	FeS/FeS	Zirc-4/FeS
Coolant	LiPb	He
Trit. Breeder	LiPb	LiO <sub>2</sub>
Reflector	FeS, LiPb	FeS, He
Shield	FeS, LiPb, B <sub>4</sub> C, ZrD <sub>2</sub> , W	W, B <sub>4</sub> C, He
Magnets	NbSn, NbTi/He (OFHC/LN <sub>2</sub> )	NbSn, NbTi/He
First-Wall	Be-coated FeS, LiPb	Be-coated FeS, He
Divertor	W-tiles on Cu-CuCrZr, LiPb	W-tiles on Cu – CuCrZr, He

A series of design studies was performed for the FTWR. The objectives of the original FTWR study<sup>18</sup> were to achieve minimum size by using liquid nitrogen cooled Cu magnets, to achieve electrical power breakeven ( $Q_e = 1$ ), and to achieve an adequate transmutation rate to dispose of the spent

nuclear fuel being generated by three 1000 MW<sub>e</sub> LWRs. The second FTWR-SC study<sup>19</sup> was a modification of the FTWR design to replace the Cu magnets with superconducting magnets and to provide enough shielding to make them lifetime components. The core radius became larger as a

result, and the power density was held constant so that the FTWR nuclear and thermal core design<sup>18,24</sup> and fuel cycle analysis<sup>18,23</sup> could be simply scaled up by volume. The third FTWR-AT study<sup>20</sup> investigated the reduction in size that could be achieved in a superconducting design by using advanced tokamak physics; again the same core power density was used.

The GCFTR series of studies is now in progress. The objectives of the first GCFTR study<sup>22</sup> were to achieve > 90% burnup of transuranics in the coated fuel particles without reprocessing the coated TRU pellets, achieve an adequate transmutation rate to dispose of the spent nuclear fuel being generated by three 1000 MW<sub>e</sub> LWRs, and to achieve net

electric power while avoiding the very high temperatures (and associated materials requirements) characteristic of other gas-cooled reactor designs. During the later stages of the GCFTR study it became apparent that the superconducting magnet thicknesses could be reduced, and the preliminary GCFTR-2 study was performed to assess the effect on the design.

The major dimensions of the various design concepts are given in Table II. The plasma-related parameters for the FTWR and GCFTR designs are given in Table III.

**TABLE II Dimensions (m) of FTWR and GCFTR Designs**

Parameter	FTWR <sup>a</sup>	FTWR-SC <sup>b</sup>	FTWR-AT <sup>c</sup>	GCFTR <sup>d</sup>	GCFTR-2 <sup>d</sup>
Major Radius <sup>e</sup> , R <sub>0</sub>	3.10	4.50	3.86	4.15	3.70
Fluxcore, R <sub>fc</sub>	1.24	1.10	0.65	0.66	0.66
CS+TF, Δ <sub>mag</sub>	0.57	1.68	1.20	1.50	1.13
Refl+Shld, Δ <sub>rs</sub>	0.40	0.65	0.90	0.86	0.82
Plasma, a <sub>plasma</sub>	0.89	0.90	1.10	1.04	1.08
Core					
Inner Radius, R <sub>in</sub>	4.00	5.40	5.00	5.25	4.84
Width, W	0.40	0.40	0.40	1.12	1.12
Height, H	2.28	2.28	2.28	3.00	3.00

<sup>a</sup>ITER physics, LN<sub>2</sub> Cu magnets, PbLi/TRU-metal reactor<sup>18</sup>; <sup>b</sup>ITER physics, ITER SC magnets, PbLi/TRU-metal reactor<sup>19</sup>; <sup>c</sup>AT physics, SC magnets, PbLi/TRU-metal reactor<sup>20</sup>; <sup>d</sup>ITER physics, ITER SC magnets, He/TRU-TRISO reactor<sup>22</sup>; <sup>e</sup> includes gap, first-wall, scrape-off layer and items below.

The requirements on β<sub>N</sub> and confinement are within the range routinely achieved in present experiments, and the requirements on β<sub>N</sub>, confinement, energy amplification Q<sub>p</sub>, and fusion power level are at or below the ITER level. The requirement on the current-drive efficiency, after calculation of bootstrap current fraction using ITER scaling, is only somewhat beyond what has been achieved to date (γ<sub>CD</sub> = 0.45 in JET and 0.35 in JT60-U). The ongoing worldwide tokamak program is addressing the current-drive/bootstrap current/steady-state physics issue. The current-drive efficiency/bootstrap fraction needed for FTWR/GCFTR is certainly within the range envisioned for Advanced Tokamak operation and may be achieved in ITER.

### III. TRANSMUTATION REACTOR CORES

#### III.A. FTWR

The fuel is a transuranic zirconium alloy (TRU-10Zr) dispersed in a zirconium matrix and clad with a ferritic steel similar to HT-9. The relative amounts of transuranics and zirconium in the fuel

region are adjusted to achieve the desired neutron multiplication (k<sub>eff</sub> = 0.95) at the beginning of each cycle. At equilibrium, the transuranics will constitute approximately 45% of the fuel volume. The annular transmutation reactor core is outboard of the plasma, and both are surrounded by reflector and shield (Fig. 1). The design of the FTWR transmutation reactor is based on the ANL ATW blanket design studies<sup>8</sup>. The same pin and assembly geometry was used, with the exception that the length of the assembly was increased to 228 cm. Table IV gives the basic data for the fuel assembly design. The reactor core is 40 cm thick and consists of 470 assemblies, 1/5 of which will be ‘half assemblies’ placed in the gaps along the interior and exterior surfaces of the reactor region to produce a more uniform annular distribution, as shown in Fig. 2.

The total coolant mass flow rate required to maintain T<sub>in</sub> = 548 K and T<sub>out</sub> = 848 K is 51630 kg/s. The required pumping power is 130 MW, the majority of which is needed to overcome MHD losses.

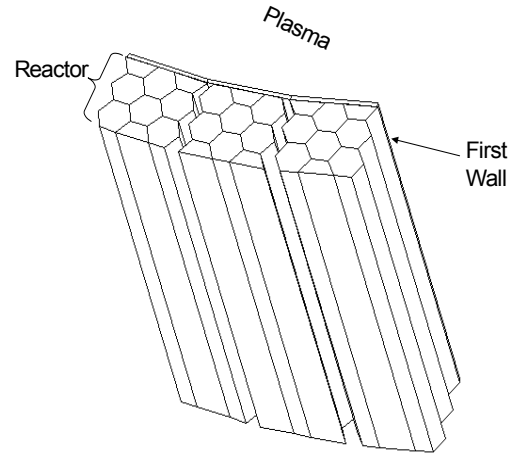
**TABLE III Tokamak Neutron Source Parameters for Transmutation Reactors**

Parameter	FTWR <sup>a</sup>	FTWR-SC <sup>b</sup>	FTWR-AT <sup>c</sup>	GCFTR <sup>d</sup>	GCFTR-2 <sup>d</sup>	ITER <sup>e</sup>
Fusion power, $P_{fus}$ (MW)	$\leq 150$	$\leq 225$	$\leq 500$	$\leq 180$	$\leq 180$	410
Neutron source, $S_{fus}(10^{19} \text{ \#}/s)$	$\leq 5.3$	$\leq 8.0$	$\leq 17.6$	$\leq 7.1$	$\leq 7.1$	14.4
Major radius, R (m)	3.1	4.5	3.9	4.2	3.7	6.2
Aspect ratio, A	3.5	5.0	3.5	4.0	3.4	3.1
Elongation, $\kappa$	1.7	1.8	1.7	1.7	1.7	1.8
Current, I (MA)	7.0	6.0	8.0	7.2	8.3	15.0
Magnetic field, B (T)	6.1	7.5	5.7	6.3	5.7	5.3
Safety factor, $q_{95}$	3.0	3.1	3.0	3.0	3.0	
Confinement, $H_{IPB98}(y,2)$	1.1	1.0	1.5	1.0	1.0	1.0
Normalized beta, $\beta_N$	$\leq 2.5$	$\leq 2.5$	4.0	2.0	2.0	1.8
Plasma Power Mult., $Q_p$	$\leq 2.0$	$\leq 2.0$	4.0	2.9	3.1	10
CD efficiency, $\gamma_{cd}(10^{-20} \text{ A/Wm}^2)$	0.37 <sup>f</sup>	0.23	0.04	0.5	0.61	
Bootstrap current fraction, $f_{bs}$	0.40 <sup>f</sup>	0.50	$\geq 0.90$	0.35	0.31	
Neut. flux, $\Gamma_n(\text{MW/m}^2)$	$\leq 0.8$	$\leq 0.8$	$\leq 1.7$	$\leq 0.9$	$\leq 0.6$	0.5
Heat flux, $q_{fw}(\text{MW/m}^2)$	$\leq 0.34$	$\leq 0.29$	$\leq 0.5$	$\leq 0.23$	$\leq 0.23$	0.15
Availability (%)	$\geq 50$	$\geq 50$	$\geq 50$	$\geq 50$	$\geq 50$	

<sup>a-d</sup> same as Table II; <sup>e</sup> ITER design parameters. (Ref. 25); <sup>f</sup> bootstrap current calc. using ITER scaling, then required CD effc. calculated.

**TABLE IV FTWR Fuel Assembly Design**

Pin Diameter (cm)	0.635
Clad thickness (cm)	0.05588
Pitch	Triangular
Pitch to Diameter	1.727
Pins per assembly	217
Structure Pins	7
Fuel Smear density	85%
Hexagonal Assembly Pitch	16.1
Assembly Length (cm)	228
Assemblies	470
Pumping Power (MW)	130
Volume %	
Fuel	17.01
Structure	10.44
Coolant	69.55
Materials	
Fuel	TRU-10Zr/Zr
Structure	FeS
Coolant	Li17Pb83

**Fig. 2 FTWR Transmutation Reactor Core Configuration Outboard of Plasma Chamber**

### III.B. GCFTR

Design concepts were developed for a TRISO (tri-material isotropic) particle and for a BISO (bi-material isotropic) particle, as shown in Fig. 3.

The TRISO particle has a TRU kernel (300  $\mu\text{m}$  diameter) surrounded by a 50% porous buffer layer (100  $\mu\text{m}$ ) of ZrC to allow for fission product recoil

and to accommodate fission product gas buildup, followed by a structural layer (20  $\mu\text{m}$ ) of pyrolytic carbon which prevents chlorine attack of the kernel during the coating process and contains the fission products, followed by a structural layer (25  $\mu\text{m}$ ) of SiC which shrinks under irradiation to provide an inward pressure to counteract the fission product gas pressure buildup, followed by an outer pyrolytic carbon layer (35  $\mu\text{m}$ ) to prevent interaction of the SiC

with any metallic cladding material. The BISO particle has a similar kernel and buffer layer followed

by a (25  $\mu\text{m}$ ) pyrolytic carbon structural layer and then by a (35  $\mu\text{m}$ ) ZrC structural outer layer.

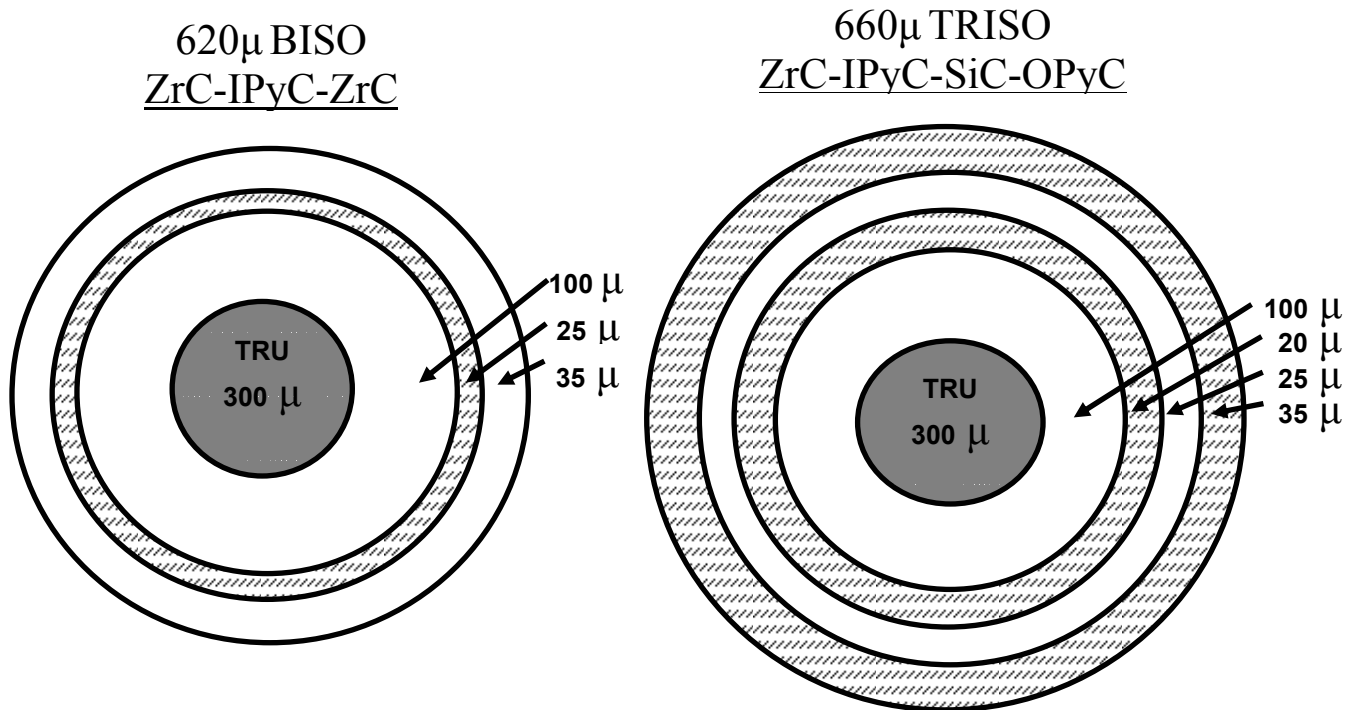


Figure 3: BISO and TRISO coated fuel particles

It is an objective to achieve very high burnup before loss of integrity of the coated fuel particle or degradation of fission product gas containment becomes unacceptable. The TRISO and BISO particles are predicted to reach 155 MPa at 90% FIMA and 180 MPa at 99% FIMA for the maximum predicted fuel centerline temperature of 560 °C. The operational pressure limit due to the compressive yield strength of SiC for the TRISO particle is 345 MPa, and the similar limit for the BISO particle is 352 MPa. These limits correspond to fuel centerline temperature limits of 1700 and 1520 °C at 90% FIMA and 99% FIMA, respectively, for the BISO particle; and to fuel centerline temperature limits of 1690 and 1510 °C at 90% FIMA and 99% FIMA, respectively, for the TRISO particle.

A thermal analysis was performed for Zirc-4 clad pins in which the BISO fuel particles were uniformly homogenized in the Zirc-4 matrix material. A configuration with 207200 fuel pins 0.60 cm in radius with a gap of 0.005 cm and a 0.057 cm thick cladding was chosen for the analysis. For 3000 MW<sub>th</sub> total reactor power uniformly distributed in the fuel pins, the volumetric heat source is  $q''' = 42.2 \text{ MW/m}^3$ . With a He mass flow rate of 2870 kg/s, the He coolant entered at 280 C and exited at 481 C, the

maximum clad temperature was 513 C (well below the 1845 °C m. p. for Zircaloy), the maximum homogenized fuel centerline temperature was 560 C, well below the 2000+ °C melting point for TRU-oxides, and the He pumping power was 0.15 MW. A He coolant v/o  $\geq 25\%$  would be adequate for heat removal under normal operating conditions.

#### IV. FUEL SEPARATION AND FABRICATION SYSTEMS

##### IV.A. FTWR

The processing system for the FTWR will be identical to the waste processing system being developed for the ATW system<sup>57</sup>. The waste processing system consists of three basic components. The first is a uranium extraction system (UREX) that will separate the bulk uranium and fission products in the SNF from the transuranic elements. The transuranic elements and the rare earth fission products will then be transferred to a pyrometallurgical system (Pyro-A) that will separate the rare earths from the transuranic elements and convert the latter to a metallic form for fuel manufacturing. The discharged FTWR fuel will be sent to a separate

pyro-metallurgical system (Pyro-B) where the residual actinides will be recovered. The recovered materials from Pyro-A and Pyro-B will be blended together and manufactured into new fuel elements for the FTWR.

The UREX system is assumed to remove 99.995% of the uranium and all of the fission products that are not rare earth elements. The Pyro A system is assumed to remove 95% of the rare earth fission products and recover 99.9% of the transuranic elements. The Pyro B system is assumed to remove 95% of the rare earth fission products, remove 100% of all other fission products, and recover 99.9% of the transuranic elements. In addition to the recovery fractions, the total fraction of transuranics that end up in the waste stream is a strong function of fractional burnup achieved during each residence in the FTWR. For the FTWR, each MTU of SNF will result in 70 g of transuranics in the waste stream.

#### IV.B. GCFTR

Aqueous systems for separating the TRU in LWR spent nuclear fuel (SNF) and systems for fabricating it into coated particle fuel have been identified for the GCFTR. The uranium (99.995%) is first removed from the SNF using a UREX process. The remaining 0.005% of the uranium, the TRU and the fission products are then treated with a TRUEX process and a TRU/lanthanide separation step to remove virtually all of the fission products, which are sent to a high-level waste repository. The TRU emerging from the TRUEX process (including 0.005% of the uranium and virtually all of the transuranics) is then fabricated into coated TRU fuel particles. The heavy metal composition of the 'TRU' emerging from this process is (U—0.43%, Np—4.32%, Pu—84.91%, Am—10.21%, Cm—0.13%).

The fabrication process starts with evaporation of the TRU stream, which is then passed through a calciner to form a mixture of transuranic oxides. Finally, a ZrC buffer layer and the pyrolytic carbon and ZrC (BISO) or pyrolytic carbon and SiC (TRISO) layers are coated onto the particles. Less than 0.1% TRU loss is assumed during the fabrication process.

### V. FUEL CYCLE ANALYSIS

The composition changes in the fuel cycle were calculated with the REBUS fuel cycle code<sup>28</sup>

#### V.A. FTWR

The reference fuel cycle assumes that the FTWR fuel will remain in the reactor for 4 cycles of

623 days each and then be reprocessed, blended with 'fresh' SNF and fabricated into new fuel elements for re-insertion into a FTWR. A beginning-of-cycle (BOC) transuranic loading of 27 MTU will produce  $k_{\text{eff}}=0.95$ , the largest value during the cycle. Over the 40 FPY plant life of the first generation of FTWRs, the original charge of LWR feed will be reprocessed 5 times.

The initial charge of the reactor and the first reload batch will require approximately 3500 MTU of LWR SNF to manufacture these fuel elements. Following this, approximately 190 MTU of LWR SNF will be processed in each subsequent 623-day cycle. A first generation FTWR will process approximately 74 MT of transuranics from LWR SNF, of which approximately 56% will be fissioned, 0.2% will be lost to the waste streams, and 44% will be used in a second generation FTWR.

The second and subsequent generations of FTWRs will use the fuel from the previous generation FTWRs and therefore operate in the equilibrium mode over their entire life. Repeated recycling of the discharged transuranics from FTWRs in successive generations of FTWRs will ultimately result in the destruction of 99.4% of the transuranics discharged from LWRs.

#### V.B. GCFTR

An emphasis in the GCFTR investigation was achieving sufficiently high (>90% FIMA) TRU burnup that the coated fuel particles can be burned and then removed from the reactor and directly deposited in a waste repository without the necessity of reprocessing. To this end, we again examined a multi-batch fuel cycle in which the reactivity decrease (from  $k = 0.95$  at BOC) associated with fuel burnup was partially offset by an increase in neutron source strength over the burn cycle.

For the reference 5-batch, 600 day burn cycle, 8.2 year fuel cycle, the BOC TRU loading was 36 MT for the TRISO fuel and 47 MT for the BISO fuel. For both fuels, the BOC  $k_{\text{eff}} = 0.95$  and neutron source  $P_{\text{fusion}} \approx 40$  MW, and the end of cycle  $k_{\text{eff}} \approx 0.81$  (0.87 for BISO) and neutron source  $P_{\text{fusion}} \approx 170$  MW (107 MW for BISO). About 23% of the BOC TRU loading is fissioned in an 8.2 year fuel cycle. The fuel would have to be resident in the core for about 10 such fuel cycles to achieve 90% TRU burnup.

#### V.C. Performance

The FTWR and GCFTR cores are designed to operate at a nominal fission power level of 3000 MW<sub>th</sub>, which corresponds to the fission of 1.1 metric

tons of TRU per EFY. A typical 1000 MW<sub>e</sub> LWR produces 0.36 metric tons of TRU per EFY. Hence, one FTWR or GCFTR would be able to ‘support’ (burn the TRU discharged from) three 1000 MW<sub>e</sub> LWRs.

The FTWR and GCFTR also produce electrical power. The original FTWR with Cu magnets has a large ohmic heat removal power requirement and was designed for electrical breakeven, but superconducting FTWRs would produce net electrical power, as the GCFTR does. Using a Brayton cycle with 32% thermal-to-electrical energy conversion efficiency to convert the 3000 MW thermal power, the gross electric power production of a GCFTR would be 1024 MW<sub>e</sub>. The electrical power requirements for the operation of the GCFTR are 305 MW<sub>e</sub>, leading to an electric power amplification factor of  $Q_e = 3.4$  and a net electric power production of 719 MW<sub>e</sub>.

#### V.D. Deployment

Availability of the transmutation reactor will determine the annual transmutation rate, hence the number of transmutation reactors needed to service the USA LWR fleet. The projected SNF transmutation rate is 100A MTU per year for both the FTWR and the GCFTR, where A is the availability. (The other design variants with somewhat higher power would have somewhat higher transmutation rates.) At the present level of nuclear power production in the US, about 100 LWRs produce about 2000 MTU of SNF per year. Thus, 20/A transmutation reactors would be needed to handle the annual SNF production, assuming the present level of nuclear power continues indefinitely. Operating at 50% availability, 40 sub-critical reactors would accomplish this transmutation mission. At 75% availability, only 25 would be needed.

### VI. COMPONENT LIFETIMES

The design lifetime of the GCFTR is 40 years at 75% availability, or 30 EFY. The magnet systems, shields, reflectors, etc. are designed as lifetime components. However, the reactor fuel and structure, the first-wall of the plasma chamber and the divertor will have to be replaced one or more times over the 30 EFY because of radiation damage.

It is envisioned that the coated fuel pellets will be imbedded in a matrix material and clad in Zircalloy-4 fuel elements and arranged in fuel assemblies constructed of ferritic steel. The fuel elements will be left in the reactor for five consecutive 600 EFY cycles, which requires that the clad not fail in this “residence” time, during which it

will accumulate a fast ( $E > 0.1$  MeV) neutron fluence of  $4.2 \times 10^{22}$  n/cm<sup>2</sup>. We have not been able to determine the radiation damage lifetime of Zircalloy-4, but it is widely used as cladding in nuclear reactors.

The structural material of the fuel assembly will accumulate a fast neutron fluence of  $1.9 \times 10^{23}$  n/cm<sup>2</sup> over the 30 EFY design lifetime. The estimated<sup>29</sup> radiation damage lifetime of ferritic steel is 80-150 dpa, or  $1.5$ - $3.0 \times 10^{23}$  n/cm<sup>2</sup>, implying that the core fuel assembly structure may need to be replaced once over the 30 EFY lifetime of the GCFTR.

When the fuel is removed from the reactor after its residence time, the cladding will be replaced, and the matrix material (SiC or Zircalloy-4) will be replaced if necessary, but the coated fuel pellets will be blended with “fresh” fuel pellets and re-fabricated into fuel elements to be re-inserted into another GCFTR. The objective is to repeatedly recycle the fuel pellets until they reach  $> 90\%$  FIMA, without reprocessing. The fast neutron fluence will be  $4.1 \times 10^{23}$  and  $8.2 \times 10^{23}$  n/cm<sup>2</sup> at 90% and 99% FIMA, respectively. A fluence lifetime in this range is then a non-trivial requirement of the coated particle fuel development program.

The first-wall of the plasma chamber and the plasma-facing part of the divertor will accumulate fast neutron fluences of 7.5 and  $5.8 \times 10^{23}$  n/cm<sup>2</sup>, respectively, over the 30 EFY lifetime of the GCFTR. The radiation damage limit of the ferritic steel first-wall structure is  $1.5$ - $3.0 \times 10^{23}$  n/cm<sup>2</sup>, which implies that it will be necessary to replace the first-wall 2-4 times over the 30 EFY lifetime of the GCFTR. Erosion of the divertor by the incident plasma ion flux will necessitate several replacements over the 30 EFY lifetime of the GCFTR.

The superconducting magnets are shielded to reduce the fast neutron fluence to the superconductor and the rad dose to the insulators below their respective limits— $10^{19}$  n/cm<sup>2</sup> fast neutron fluence for Nb<sup>3</sup>Sn and  $10^9$  rads for organic insulators ( $10^{10}$  rads for ceramic insulators).

### VII. TRANSMUTATION MISSION IN THE FUSION PROGRAM

A transmutation reactor can be driven by a tokamak fusion neutron source based on physics ( $H$ ,  $\beta_N$ ,  $Q_p$ , etc.) similar to or less demanding than that used for the ITER design, except for the need to achieve a higher bootstrap current fraction and/or higher current drive efficiency. This tokamak neutron source can be constructed with the fusion technology being developed for ITER, but will need to achieve greater availability, hence have greater

component reliability, than ITER. Achieving higher availability, which will require various component test facilities, must be addressed in the fusion development program, but would have a higher near-term priority if the transmutation mission were undertaken.

The reactor technology for the sub-critical reactor driven by the fusion neutron source would be adapted from the reactor (nuclear, fuel, cooling, separations, materials) technologies being developed in the nuclear program (e.g. GEN-IV, AFCI, NGNP), but these technologies must be modified to provide for the tritium breeding requirement. A fusion nuclear technology program would have to be

revived with this goal. There is a need to develop a long-lived structural material, primarily for the fuel assemblies of the sub-critical reactor, but also for the first wall of the fusion neutron source.

The technical requirements for a tokamak fusion neutron source that would fulfill the transmutation mission are significantly less demanding than for an economically competitive tokamak electrical power reactor and somewhat less demanding than for a DEMO, as indicated in Table V.

**Table V Requirements for a Tokamak Neutron Source, Electric Power Reactor and DEMO**

Parameter	Transmutation	Electric Power <sup>a</sup>	DEMO <sup>b</sup>
Confinement $H_{IPB98}(Y,2)$	1.0	1.5-2.0	1.5-2.0
Beta $\beta_N$	< 2.5	> 5.0	> 4.0
Power Amplification $Q_p$	< 3	> 25	> 10
Bootstrap Current Fraction $f_{bs}$	0.2-0.5	0.9	0.7
Neutron wall load (MW/m <sup>2</sup> )	$\leq 1.0$	> 4.0	> 2.0
Fusion Power (MW)	$\leq 200$	3000	1000
Pulse length/duty factor	long/steady-state	long/steady-state	long/steady-state
Availability (%)	> 50	90	< 50

<sup>a</sup> ARIES studies (Ref. 30); <sup>b</sup> DEMO studies (Ref. 31)

## VIII. CONCLUSIONS

A sub-critical transmutation reactor, based on adaptation of nuclear and separations technology presently being developed in the DoE Nuclear Energy Program to accommodate tritium breeding, and driven by a tokamak D-T fusion neutron source, based on the physics and technology presently being developed in the DoE Fusion Energy Sciences Program, could be online in 2040. The tokamak neutron source, which would be about  $R = 4$  m in major radius and produce < 200 MW of D-T fusion power, could be designed on the basis of the existing plasma physics and fusion technology databases, with only a few modest extensions. The pacing items for the neutron source would be operation of a prototype plasma (e.g. ITER) experiment and component test facilities to gain the experience necessary to achieve > 50% availability in operation of the fusion neutron source.

**ACKNOWLEDGMENT** This work was supported in part by DoE grant DE-FG0296ER54350. The contributions of Profs. D. Tedder, J. Lackey and C. de Oliveira and of the students in the Georgia Tech NRE design projects to the development of the material in this paper are gratefully acknowledged.

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